TXU Power

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Ref: #10CFR50.73(a)(2)(iv)(A)

CPSES-200602239
Log # TXX-06186

December 18, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-446
ACTUATION OF REACTOR PROTECTION SYSTEM
LICENSEE EVENT REPORT 446/06-003-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 06-003-00 for Comanche Peak Steam Electric Station Unit 2, "Reactor Trip Due to Feedwater Regulating Valve Malfunction."

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2.

IE22

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W. Madden
Director, Oversight and Regulatory Affairs

GLM
Attachment

c - B. S. Mallett, Region IV
M. C. Thadani, NRR
Resident Inspectors, CPSES

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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Title (4)
Reactor Trip Due to Feedwater Regulating Valve Malfunction

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Name	Docket Numbers
									N/A	05000
10	29	2006	2006	003	00	12	18	06		05000

Operating Mode (9): 1

Power Level (10): 80%

This report is submitted pursuant to the requirements of 10 CFR : (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	20.2201(d)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
	20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
	20.2203(a)(2)(i)		50.36(c)(2)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(c)(1)(ii)(A)	X	50.73(a)(2)(iv)(A)		50.72(a)(2)(x)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
	20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
	20.2203(a)(2)(v)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(C)		OTHER
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

Licensee Contact For This LER (12)

Name	Telephone Number (Include Area Code)
Tim Hope – Regulatory Performance Manager	(254) 897-6370

Complete One Line For Each Component Failure Described in This Report (13)

Cause	System	Component	Manufacturer	Reportable To EPIX	Cause	System	Component	Manufacturer	Reportable To EPIX

Supplemental Report Expected (14)

				EXPECTED SUBMISSION DATE (15)	Month	Day	Year
	YES (If YES, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 29, 2006, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1 operating at approximately 80% power following the completion of the ninth refueling outage. While holding for xenon stabilization in preparation for incore/excore calibration, a "Steam Generator 3 Steam and Feedwater Flow Mismatch" was received. Manual control of the Steam Generator (SG) 3 Feedwater regulating valve was taken, but operators were unable to control feed flow. A manual reactor trip was initiated in anticipation of an automatic reactor trip on SG3 Lo Lo level. Auxiliary Feedwater automatically started on low-low level in SG 3. All systems responded normally during and following the trip.

The cause of this event was believed to be a loose wire on a Unit 2 Feedwater regulating valve Weidmuller terminal block. Corrective actions included tightening the loose wire and inspecting the connections in other safety related Unit 2 Feedwater regulating valve and bypass valve terminal boxes. Future actions include inspecting the connections for safety related Unit 1 Feedwater regulating valve and bypass valve terminal boxes and checking the tightness of both the instrument and field terminals on all Weidmuller terminal blocks whenever either side of a circuit is being worked on a Weidmuller terminal block.

All times in this report are approximate and Central Time unless noted otherwise.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

10CFR50.73(a)(2)(iv)(A); "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B)."

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On October 29, 2006, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1, operating at 80% power following completion of the ninth refueling outage.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems, or components that contributed directly to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On October 29, 2006 Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1 operating at approximately 80% power following the completion of the ninth refueling outage. At 1518 hours, while holding for xenon stabilization in preparation for an incore/excore calibration, a "Steam Generator 3 Steam and Feedwater Flow Mismatch" alarm was received. The Unit 2 Balance Of Plant operator (utility, licensed) took manual control of the Steam Generator (SG) 3 Main Feedwater (MFW) flow control valve and raised demand to match feed flow and steam flow. After the operator raised the feed flow demand at the SG3 MFW flow control valve, feed flow began to rise and actually exceeded steam flow. The feed flow demand at the SG3 MFW flow control valve was then reduced to lower feed flow to match the steam flow when feed flow dropped off drastically. Demand was once again raised (to 100%) but feed flow continued to lower. The Unit Supervisor (utility, licensed) ordered a reactor trip at 1520 hours due to SG3 level lowering uncontrollably. SG3 level was approximately 40% at the time of the trip and lowering rapidly. Auxiliary feedwater automatically started as expected due to Lo Lo level in SG3. All systems responded normally during and following the trip and the unit was stabilized in Mode 3.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE, OR PROCEDURAL OR PERSONNEL ERROR

Operators (utility, licensed) in the Unit 2 Control Room received a "Steam Generator 3 Steam and Feedwater Flow Mismatch" alarm.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

Not applicable – there were no component failures associated with this event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable – there were no component failures associated with this event.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - there were no component failures associated with this event.

D. FAILED COMPONENT INFORMATION

Not applicable - there were no component failures associated with this event.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Both Motor Driven Auxiliary Feedwater Pumps (AFW) and the Turbine Driven Auxiliary Feedwater Pump started.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not applicable – there was no safety system train inoperability that resulted from this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

A loss of normal Feedwater resulting from pump failure, valve malfunction, or loss of offsite power leads to a reduction in the capability of the secondary system to remove heat generated in the reactor core. These events are analyzed in section 15.2.7 of the CPSES Updated Final Safety Analysis Report (UFSAR) which uses conservative assumptions in the analysis to minimize the energy removal capability of the Auxiliary Feedwater system.

The October 29, 2006 event occurred with the reactor at approximately 80 percent power. All systems and components functioned as designed. The event is bounded by the UFSAR accident analysis which assumes an initial power level of 102 percent and the worst single failure in the Auxiliary Feedwater system for a loss of Feedwater event. There were no safety system functional failures associated with this event. The UFSAR analysis shows that a loss of normal Feedwater does not adversely affect the core, the reactor coolant systems, or the steam system; therefore, this event posed no threat to the health and safety of the public.

Based on the above, it is concluded that the health and safety of the public was unaffected by this condition and this event has been evaluated to not meet the definition of a safety system functional failure per 10CFR50.73(a)(2)(v).

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IV. CAUSE OF THE EVENT

The cause of this event was believed to be a loose wire on the SG3 Feedwater regulating valve Weidmuller terminal block. The loose wire in the Feedwater regulating valve control circuit caused a high resistance connection and voltage drop that caused the solenoid valve in the pneumatic control system to vent air while still supplying air from the positioner. This loss of air caused a loss of control and closure of the Feedwater regulating valve. The loose connection was most likely the result of poor workmanship during initial installation of the field cable (believed to be installed during Unit 2 construction).

In 1987, prior to the startup of Units 1 and 2, loose connections in Weidmuller terminals blocks were identified. As part of the corrective actions for Unit 1, the safety related terminals were inspected and tightened as required. Due to the stage of Unit 2 construction, the Unit 2 corrective actions involved procedure changes to verify tightness. Based on the estimated time of the cable installation, the affected cable's tightness should have been verified under the improved termination procedures. However, it could not be specifically determined how the affected cable became loose.

V. CORRECTIVE ACTIONS

Corrective actions included tightening the loose wire, checking the connections associated with all four Unit 2 Feedwater regulating valves and all four Unit 2 Feedwater regulating bypass valves for tightness. Although some lack of full tightness conditions were identified, the associated design functions would have been performed for all of these conditions.

As a part of the CPSES Corrective Action Program, the tightness of Weidmuller terminals in Unit 1 Feedwater regulating and bypass valve terminal boxes will be verified. Additionally, maintenance procedures will be revised to require a check of tightness of both the instrument and field terminals -whenever either side of a circuit is being worked on a Weidmuller terminal block.

VI. PREVIOUS SIMILAR EVENTS

There have been no previous similar reportable events at CPSES in the last three years.